to comply associated with a substantial safety hazard has been previously reported under part 21 of this chapter or under § 73.71 of this chapter under § 50.55(e) or § 50.73 of this part. For holders of construction permits issued prior to October 29, 1991. Evaluation, reporting and recordkeeping requirements of § 50.55(e) may be met by complying with the comparable requirements of part 21 of this chapter.

(f)(1) Each nuclear power plant or fuel reprocessing plant construction permit holder subject to the quality assurance criteria in appendix B of this part shall implement, pursuant to § 50.34(a)(7) of this part, the quality assurance program described or referenced in the Safety Analysis Report, including changes to that report.

(2) Each construction permit holder described in paragraph (f)(1) of this section shall, by June 10, 1983, submit to the appropriate NRC Regional Office shown in appendix D of part 20 of this chapter the current description of the quality assurance program it is implementing for inclusion in the Safety Analysis Report, unless there are no changes to the description previously accepted by NRC. This submittal must identify changes made to the quality assurance program description since the description was submitted to NRC. (Should a permit holder need additional time beyond June 10, 1983 to submit its current quality assurance program description to NRC, it shall notify the appropriate NRC Regional Office in writing, explain why additional time is needed, and provide a schedule for NRC approval showing when its current quality assurance program description will be submitted.)

(3) After March 11, 1983, each construction permit holder described in paragraph (f)(1) of this section may make a change to a previously accepted quality assurance program description included or referenced in the Safety Analysis Report, provided the change does not reduce the commitments in the program description previously accepted by the NRC. Changes to the quality assurance program description that do not reduce the commitments must be submitted to NRC within 90 days. Changes to the quality assurance program description that do reduce the commitments must be submitted to NRC and receive NRC approval before implementation, as follows:

(i) Changes to the Safety Analysis Report must be submitted for review as specified in § 50.4. Changes made to NRC-accepted quality assurance topical report descriptions must be submitted as specified in § 50.4.

(ii) The submittal of a change to the Safety Analysis Report quality assurance program description must include all pages affected by that change and must be accompanied by a forwarding letter identifying the change, the reason for the change, and the basis for concluding that the revised program incorporating the change continues to satisfy the criteria of appendix B of this part and the Safety Analysis Report quality assurance program description commitments previously accepted by the NRC (the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items).

(iii) A copy of the forwarding letter identifying the changes must be maintained as a facility record for three years.

(iv) Changes to the quality assurance program description included or referenced in the Safety Analysis Report shall be regarded as accepted by the Commission upon receipt of a letter to this effect from the appropriate reviewing office of the Commission or 60 days after submittal to the Commission, whichever occurs first.

[21 FR 355, Jan. 19, 1956, as amended at 32 FR 4055, Mar. 15, 1967; 35 FR 11461, July 17, 1970; 35 FR 19661, Dec. 29, 1970; 36 FR 11424, June 12, 1971; 37 FR 6460, Mar. 30, 1972; 38 FR 1272, Jan. 11, 1973; 41 FR 16446, Apr. 19, 1976; 42 FR 43385, Aug. 29, 1977; 48 FR 1029, Jan. 10, 1983; 51 FR 40309, Nov. 6, 1986; 56 FR 36091, July 31, 1991; 59 FR 14087, Mar. 25, 1994; 68 FR 58809, Oct. 10, 2003]

§ 50.55a Codes and standards.

Each operating license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g) of this section and each construction permit for a utilization facility is subject to the following conditions in addition to those specified in § 50.55.

(a)(1) Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

(2) Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code specified in paragraphs (b), (c), (d), (e), (f), and (g) of this section. Protection systems of nuclear power reactors of all types must meet the requirements specified in paragraph (h) of this section.

(3) Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

(b) The ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants, which are referenced in paragraphs (b)(1), (b)(2), and (b)(3) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51. NRC Regulatory Guide 1.84, Revision 33, ''Design, Fabrication, and Materials Code Case Acceptability, ASME Section III'' (August 2005); NRC Regulatory Guide 1.147 (Revision 0-February 1981), including Revision 1 through Revision 14 (August 2005), ''Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1''; and Regulatory Guide 1.192, ''Operation and Maintenance Code Case Acceptability, ASME OM Code'' (June 2003), have been approved for incorporation by reference by the Director of the Office of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51. These regulatory guides list ASME Code cases which the NRC has approved in accordance with the requirements in paragraphs (b)(4), (b)(5), and (b)(6). Copies of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants may be purchased from the American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016. Single copies of NRC Regulatory Guides 1.84, Revision 33; 1.147, Revision 14; and 1.192 may be obtained free of charge by writing the Reproduction and Distribution Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; or by fax to 301–415–2289; or by e-mail to *DISTRIBUTION@nrc.gov*. Copies of the ASME Codes and NRC Regulatory Guides incorporated by reference in this section may be inspected at the NRC Technical Library, Two White Flint North, 11545 Rockville Pike, Rockville, MD 20852–2738, or at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202–741–6030, or go to: *http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html*.

(1) As used in this section, references to Section III of the ASME *Boiler and Pressure Vessel Code* refer to Section III, and include the 1963 Edition through 1973 Winter Addenda, and the 1974 Edition (Division 1) through the 2003 Addenda (Division 1), subject to the following limitations and modifications:

(i) *Section III Materials*. When applying the 1992 Edition of Section III, licensees must apply the 1992 Edition with the 1992 Addenda of Section II of the ASME Boiler and Pressure Vessel Code.

(ii) *Weld leg dimensions*. When applying the 1989 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section, licensees may not apply paragraph NB–3683.4(c)(1), the footnote to circumferential fillet welded and socket welded joints in Figure NC–3673.2(b)–1 that permit a socket weld leg dimension to be less than 1.09 of the nominal wall thickness of the pipe or the footnote to circumferential fillet welded and socket welded joints in figure ND–3673.2(b)–1 that permit a socket weld leg dimension to be less than 1.09 of the nominal wall thickness of the pipe.

(iii) *Seismic design.* Licensees may use Articles NB–3200, NB–3600, NC–3600, and ND–3600 up to and including the 1993 Addenda, subject to the limitation specified in paragraph (b)(1)(ii) of this section. Licensees may not use these Articles in the 1994 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section.

(iv) *Quality assurance.* When applying editions and addenda later than the 1989 Edition of Section III, the requirements of NQA–1, ''Quality Assurance Requirements for Nuclear Facilities,'' 1986 Edition through the 1992 Edition, are acceptable for use provided that the edition and addenda of NQA–1 specified in NCA–4000 is used in conjunction with the administrative, quality, and technical provisions contained in the edition and addenda of Section III being used.

(v) *Independence of inspection.* Licensees may not apply NCA–4134.10(a) of Section III, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section.

(vi) *Subsection NH.* The provisions in Subsection NH, ''Class 1 Components in Elevated Temperature Service,'' 1995 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section, may only be used for the design and construction of Type 316 stainless steel pressurizer heater sleeves where service conditions do not cause the component to reach temperatures exceeding 900 °F.

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, and include the 1970 Edition through the 1976 Winter Addenda, and the 1977 Edition (Division 1) through the 2003 Addenda (Division 1), subject to the following limitations and modifications:[10]

(i) *Limitations on specific editions and addenda.* When applying the 1974 Edition, only the addenda through the Summer 1975 Addenda may be used. When applying the 1977 Edition, all of the addenda through the Summer 1978 Addenda must also be used. Addenda and editions subsequent to the Summer 1978 Addenda, that are incorporated by reference in paragraph (b)(2) of this section are not affected by these limitations.

(ii) *Pressure-retaining welds in ASME Code Class 1 piping (applies to Table IWB–2500 and IWB–2500–1 and Category B-J).* If the facility's application for a construction permit was docketed prior to July 1, 1978, the extent of examination for Code Class 1 pipe welds may be determined by the requirements of Table IWB–2500 and Table IWB–2600 Category B-J of Section XI of the ASME Code in the 1974 Edition and addenda through the Summer 1975 Addenda or other requirements the Commission may adopt.

(iii) *Steam generator tubing (modifies Article IWB–2000).* If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those in Article IWB–2000, the inservice inspection program for steam generator tubing is governed by the requirements in the technical specifications.

(iv) *Pressure-retaining welds in ASME Code Class 2 piping (applies to Tables IWC–2520 or IWC–2520–1, Category C-F).* (A) Appropriate Code Class 2 pipe welds in Residual Heat Removal Systems, Emergency Core Cooling Systems, and Containment Heat Removal Systems, must be examined. When applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of section XI of the ASME Code, the extent of examination for these systems must be determined by the requirements of paragraph IWC–1220, Table IWC–2520 Category C-F and C-G, and paragraph IWC–2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda.

(B) For a nuclear power plant whose application for a construction permit was docketed prior to July 1, 1978, when applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of section XI of the ASME Code, the extent of examination for Code Class 2 pipe welds may be determined by the requirements of paragraph IWC–1220, Table IWC–2520 Category C-F and C-G and paragraph IWC–2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda of Section XI of the ASME Code or other requirements the Commission may adopt.

(v) Evaluation procedures and acceptance criteria for austenitic piping (applies to IWB–3640). When applying the Winter 1983 Addenda and Winter 1984 Addenda, the rules of paragraph IWB–3640 may be used for all applications permitted in that paragraph, except those associated with submerged arc welds (SAW) or shielded metal arc welds (SMAW). For SAW or SMAW, use paragraph IWB–3640, as modified by the Winter 1985 Addenda.

(vi) *Effective edition and addenda of Subsection IWE and Subsection IWL, Section XI.* Licensees may use either the 1992 Edition with the 1992 Addenda or the 1995 Edition with the 1996 Addenda of Subsection IWE and Subsection IWL as modified and supplemented by the requirements in paragraphs (b)(2)(viii) and (b)(2)(ix) of this section when implementing the initial 120-month inspection interval for the containment inservice inspection requirements of this section. Successive 120-month interval updates must be implemented in accordance with paragraph (g)(4)(ii) of this section.

(vii) *Section XI References to OM Part 4, OM Part 6 and OM Part 10 (Table IWA–1600–1).* When using Table IWA–1600–1, ''Referenced Standards and Specifications,'' in the Section XI, Division 1, 1987 Addenda, 1988 Addenda, or 1989 Edition, the specified ''Revision Date or Indicator'' for ASME/ANSI OM Part 4, ASME/ANSI Part 6, and ASME/ANSI Part 10 must be the OMa–1988 Addenda to the OM–1987 Edition. These requirements have been incorporated into the OM Code which is incorporated by reference in paragraph (b)(3) of this section.

(viii) *Examination of concrete containments.* Licensees applying Subsection IWL, 1992 Edition with the 1992 Addenda, shall apply paragraphs (b)(2)(viii)(A) through (b)(2)(viii)(E) of this section. Licensees applying Subsection IWL, 1995 Edition with the 1996 Addenda, shall apply paragraphs (b)(2)(viii)(A), (b)(2)(viii)(D)(*3*), and (b)(2)(viii)(E) of this section. Licensees applying Subsection IWL, 1998 Edition through the 2000 Addenda shall apply paragraphs (b)(2)(viii)(E) and (b)(2)(viii)(F) of this section. Licensees applying Subsection IWL, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, shall apply paragraphs (b)(2)(viii)(E) through (b)(2)(viii)(G) of this section.

(A) Grease caps that are accessible must be visually examined to detect grease leakage or grease cap deformations. Grease caps must be removed for this examination when there is evidence of grease cap deformation that indicates deterioration of anchorage hardware.

(B) When evaluation of consecutive surveillances of prestressing forces for the same tendon or tendons in a group indicates a trend of prestress loss such that the tendon force(s) would be less than the minimum design prestress requirements before the next inspection interval, an evaluation must be performed and reported in the Engineering Evaluation Report as prescribed in IWL–3300.

(C) When the elongation corresponding to a specific load (adjusted for effective wires or strands) during retensioning of tendons differs by more than 10 percent from that recorded during the last measurement, an evaluation must be performed to determine whether the difference is related to wire failures or slip of wires in anchorage. A difference of more than 10 percent must be identified in the ISI Summary Report required by IWA–6000.

(D) The licensee shall report the following conditions, if they occur, in the ISI Summary Report required by IWA–6000:

(*1*) The sampled sheathing filler grease contains chemically combined water exceeding 10 percent by weight or the presence of free water;

(*2*) The absolute difference between the amount removed and the amount replaced exceeds 10 percent of the tendon net duct volume;

(*3*) Grease leakage is detected during general visual examination of the containment surface.

(E) For Class CC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report required by IWA–6000:

(*1*) A description of the type and estimated extent of degradation, and the conditions that led to the degradation;

(*2*) An evaluation of each area, and the result of the evaluation, and;

(*3*) A description of necessary corrective actions.

(F) Personnel that examine containment concrete surfaces and tendon hardware, wires, or strands must meet the qualification provisions in IWA–2300. The ''owner-defined'' personnel qualification provisions in IWL–2310(d) are not approved for use.

(G) Corrosion protection material must be restored following concrete containment post-tensioning system repair and replacement activities in accordance with the quality assurance program requirements specified in IWA–1400.

(ix) *Examination of metal containments and the liners of concrete containments.* Licensees applying Subsection IWE, 1992 Edition with the 1992 Addenda, or the 1995 Edition with the 1996 Addenda, shall satisfy the requirements of paragraphs (b)(2)(ix)(A) through (b)(2)(ix)(E) of this section. Licensees applying Subsection IWE, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, shall satisfy the requirements of paragraphs (b)(2)(ix)(A), (b)(2)(ix)(B), and (b)(2)(ix)(F) through (b)(2)(ix)(I) of this section.

(A) For Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report as required by IWA–6000:

(*1*) A description of the type and estimated extent of degradation, and the conditions that led to the degradation;

(*2*) An evaluation of each area, and the result of the evaluation, and;

(*3*) A description of necessary corrective actions.

(B) When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA–2210–1 may be extended and the minimum illumination requirements specified in Table IWA–2210–1 may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.

(C) The examinations specified in Examination Category E-B, Pressure Retaining Welds, and Examination Category E-F, Pressure Retaining Dissimilar Metal Welds, are optional.

(D) Section 50.55a(b)(2)(ix)(D) may be used as an alternative to the requirements of IWE–2430.

(*1*) If the examinations reveal flaws or areas of degradation exceeding the acceptance standards of Table IWE–3410–1, an evaluation must be performed to determine whether additional component examinations are required. For each flaw or area of degradation identified which exceeds acceptance standards, the licensee shall provide the following in the ISI Summary Report required by IWA–6000:

(*i*) A description of each flaw or area, including the extent of degradation, and the conditions that led to the degradation;

(*ii*) The acceptability of each flaw or area, and the need for additional examinations to verify that similar degradation does not exist in similar components, and;

(*iii*) A description of necessary corrective actions.

(*2*) The number and type of additional examinations to ensure detection of similar degradation in similar components.

(E) A general visual examination as required by Subsection IWE must be performed once each period.

(F) VT–1 and VT–3 examinations must be conducted in accordance with IWA–2200. Personnel conducting examinations in accordance with the VT–1 or VT–3 examination method shall be qualified in accordance with IWA–2300. The ''owner-defined'' personnel qualification provisions in IWE–2330(a) for personnel that conduct VT–1 and VT–3 examinations are not approved for use.

(G) The VT–3 examination method must be used to conduct the examinations in Items E1.12 and E1.20 of Table

IWE–2500–1, and the VT–1 examination method must be used to conduct the examination in Item E4.11 of Table IWE–2500–1. An examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE–2500–1 using the VT–3 examination method must be conducted once each interval. The ''owner-defined'' visual examination provisions in IWE–2310(a) are not approved for use for VT–1 and VT–3 examinations.

(H) Containment bolted connections that are disassembled during the scheduled performance of the examinations in Item E1.11 of Table IWE–2500–1 must be examined using the VT–3 examination method. Flaws or degradation identified during the performance of a VT–3 examination must be examined in accordance with the VT–1 examination method. The criteria in the material specification or IWB–3517.1 must be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT–3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT–3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.

(I) The ultrasonic examination acceptance standard specified in IWE–3511.3 for Class MC pressure-retaining components must also be applied to metallic liners of Class CC pressure-retaining components.

(x) *Quality Assurance.* When applying Section XI editions and addenda later than the 1989 Edition, the requirements of NQA–1, ''Quality Assurance Requirements for Nuclear Facilities,'' 1979 Addenda through the 1989 Edition, are acceptable as permitted by IWA–1400 of Section XI, if the licensee uses its 10 CFR Part 50, Appendix B, quality assurance program, in conjunction with Section XI requirements. Commitments contained in the licensee's quality assurance program description that are more stringent than those contained in NQA–1 must govern Section XI activities. Further, where NQA–1 and Section XI do not address the commitments contained in the licensee's Appendix B quality assurance program description, the commitments must be applied to Section XI activities.

(xi) *Class 1 piping.* Licensees may not apply IWB–1220, ''Components Exempt from Examination,'' of Section XI, 1989 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, and shall apply IWB–1220, 1989 Edition.

(xii) *Underwater Welding.* The provisions in IWA–4660, ''Underwater Welding,'' of Section XI, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, are not approved for use on irradiated material.

(xiii) *Mechanical clamping devices.* Licensees may use the provisions of Code Case N–523–1, ''Mechanical Clamping Devices for Class 2 and 3 Piping.'' Licensee choosing to apply Code Case N–523–1 shall apply all of its provisions.

(xiv) *Appendix VIII personnel qualification.* All personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII shall receive 8 hours of annual hands-on training on specimens that contain cracks. Licensees applying the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section may use the annual practice requirements in VII–4240 of Appendix VII of Section XI in place of the 8 hours of annual hands-on training provided that the supplemental practice is performed on material or welds that contain cracks, or by analyzing prerecorded data from material or welds that contain cracks. In either case, training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

(xv) *Appendix VIII specimen set and qualification requirements.* The following provisions may be used to modify implementation of Appendix VIII of Section XI, 1995 Edition through the 2001 Edition. Licensees choosing to apply these provisions shall apply all of the following provisions under this paragraph except for those in § 50.55a(b)(2)(xv)(F) which are optional.

(A) When applying Supplements 2, 3, and 10 to Appendix VIII, the following examination coverage criteria requirements must be used:

(*1*) Piping must be examined in two axial directions, and when examination in the circumferential direction is required, the circumferential examination must be performed in two directions, provided access is available. Dissimilar metal welds must be examined axially and circumferentially.

(*2*) Where examination from both sides is not possible, full coverage credit may be claimed from a single side for ferritic welds. Where examination from both sides is not possible on austenitic welds or dissimilar metal welds, full coverage credit from a single side may be claimed only after completing a successful single-sided Appendix VIII demonstration using flaws on the opposite side of the weld. Dissimilar metal weld qualifications must be demonstrated from the austenitic side of the weld and may be used to perform examinations from either side of the weld.

(B) The following provisions must be used in addition to the requirements of Supplement 4 to Appendix VIII:

(*1*) Paragraph 3.1, Detection acceptance criteria—Personnel are qualified for detection if the results of the performance demonstration satisfy the detection requirements of ASME Section XI, Appendix VIII, Table VIII-S4-1 and no flaw greater than 0.25 inch through wall dimension is missed.

(*2*) Paragraph 1.1(c), Detection test matrix—Flaws smaller than the 50 percent of allowable flaw size, as defined in IWB–3500, need not be included as detection flaws. For procedures applied from the inside surface, use the minimum thickness specified in the scope of the procedure to calculate a/t. For procedures applied from the outside surface, the actual thickness of the test specimen is to be used to calculate a/t.

(C) When applying Supplement 4 to Appendix VIII, the following provisions must be used:

(*1*) A depth sizing requirement of 0.15 inch RMS must be used in lieu of the requirements in Subparagraphs 3.2(a) and 3.2(c), and a length sizing requirement of 0.75 inch RMS must be used in lieu of the requirement in Subparagraph 3.2(b).

(*2*) In lieu of the location acceptance criteria requirements of Subparagraph 2.1(b), a flaw will be considered detected when reported within 1.0 inch or 10 percent of the metal path to the flaw, whichever is greater, of its true location in the X and Y directions.

(*3*) In lieu of the flaw type requirements of Subparagraph 1.1(e)(1), a minimum of 70 percent of the flaws in the detection and sizing tests shall be cracks. Notches, if used, must be limited by the following:

(*i*) Notches must be limited to the case where examinations are performed from the clad surface.

(*ii*) Notches must be semielliptical with a tip width of less than or equal to 0.010 inches.

(*iii*) Notches must be perpendicular to the surface within ±2 degrees.

(*4*) In lieu of the detection test matrix requirements in paragraphs 1.1(e)(2) and 1.1(e)(3), personnel demonstration test sets must contain a representative distribution of flaw orientations, sizes, and locations.

(D) The following provisions must be used in addition to the requirements of Supplement 6 to Appendix VIII:

(*1*) Paragraph 3.1, Detection Acceptance Criteria—Personnel are qualified for detection if:

(*i*) No surface connected flaw greater than 0.25 inch through wall has been missed.

(*ii*) No embedded flaw greater than 0.50 inch through wall has been missed.

(*2*) Paragraph 3.1, Detection Acceptance Criteria—For procedure qualification, all flaws within the scope of the procedure are detected.

(*3*) Paragraph 1.1(b) for detection and sizing test flaws and locations—Flaws smaller than the 50 percent of allowable flaw size, as defined in IWB–3500, need not be included as detection flaws. Flaws which are less than the allowable flaw size, as defined in IWB–3500, may be used as detection and sizing flaws.

(*4*) Notches are not permitted.

(E) When applying Supplement 6 to Appendix VIII, the following provisions must be used:

(*1*) A depth sizing requirement of 0.25 inch RMS must be used in lieu of the requirements of subparagraphs 3.2(a), 3.2(c)(2), and 3.2(c)(3).

(*2*) In lieu of the location acceptance criteria requirements in Subparagraph

2.1(b), a flaw will be considered detected when reported within 1.0 inch or 10 percent of the metal path to the flaw, whichever is greater, of its true location in the X and Y directions.

(*3*) In lieu of the length sizing criteria requirements of Subparagraph 3.2(b), a length sizing acceptance criteria of 0.75 inch RMS must be used.

(*4*) In lieu of the detection specimen requirements in Subparagraph 1.1(e)(1), a minimum of 55 percent of the flaws must be cracks. The remaining flaws may be cracks or fabrication type flaws, such as slag and lack of fusion. The use of notches is not allowed.

(*5*) In lieu of paragraphs 1.1(e)(2) and 1.1(e)(3) detection test matrix, personnel demonstration test sets must contain a representative distribution of flaw orientations, sizes, and locations.

(F) The following provisions may be used for personnel qualification for combined Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII qualification. Licensees choosing to apply this combined qualification shall apply all of the provisions of Supplements 4 and 6 including the following provisions:

(*1*) For detection and sizing, the total number of flaws must be at least 10. A minimum of 5 flaws shall be from Supplement 4, and a minimum of 50 percent of the flaws must be from Supplement 6. At least 50 percent of the flaws in any sizing must be cracks. Notches are not acceptable for Supplement 6.

(*2*) Examination personnel are qualified for detection and length sizing when the results of any combined performance demonstration satisfy the acceptance criteria of Supplement 4 to Appendix VIII.

(*3*) Examination personnel are qualified for depth sizing when Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII flaws are sized within the respective acceptance criteria of those supplements.

(G) When applying Supplement 4 to Appendix VIII, Supplement 6 to Appendix VIII, or combined Supplement 4 and Supplement 6 qualification, the following additional provisions must be used, and examination coverage must include:

(*1*) The clad to base metal interface, including a minimum of 15 percent T (measured from the clad to base metal interface), shall be examined from four orthogonal directions using procedures and personnel qualified in accordance with Supplement 4 to Appendix VIII.

(*2*) If the clad-to-base-metal-interface procedure demonstrates detectability of flaws with a tilt angle relative to the weld centerline of at least 45 degrees, the remainder of the examination volume is considered fully examined if coverage is obtained in one parallel and one perpendicular direction. This must be accomplished using a procedure and personnel qualified for single-side examination in accordance with Supplement 6. Subsequent examinations of this volume may be performed using examination techniques qualified for a tilt angle of at least 10 degrees.

(*3*) The examination volume not addressed by § 50.55a(b)(2)(xv)(G)(*1*) is considered fully examined if coverage is obtained in one parallel and one perpendicular direction, using a procedure and personnel qualified for single sided examination when the provisions of § 50.55a(b)(2)(xv)(G)(*2*) are met.

(H) When applying Supplement 5 to Appendix VIII, at least 50 percent of the flaws in the demonstration test set must be cracks and the maximum misorientation shall be demonstrated with cracks. Flaws in nozzles with bore diameters equal to or less than 4 inches may be notches.

(I) When applying Supplement 5, Paragraph (a), to Appendix VIII, the following provision must be used in calculating the number of permissible false calls:

(*1*) The number of false calls allowed must be D/10, with a maximum of 3, where D is the diameter of the nozzle.

(J) [Reserved]

(K) When performing nozzle-to-vessel weld examinations, the following provisions must be used when the requirements contained in Supplement 7 to Appendix VIII are applied for nozzle-to-vessel welds in conjunction with Supplement 4 to Appendix VIII, Supplement 6 to Appendix VIII, or combined Supplement 4 and Supplement 6 qualification.

(*1*) For examination of nozzle-to-vessel welds conducted from the bore, the following provisions are required to qualify the procedures, equipment, and personnel:

(*i*) For detection, a minimum of four flaws in one or more full-scale nozzle mock-ups must be added to the test set. The specimens must comply with Supplement 6, paragraph 1.1, to Appendix VIII, except for flaw locations specified in Table VIII S6–1. Flaws may be either notches, fabrication flaws or cracks. Seventy-five (75) percent of the flaws must be cracks or fabrication flaws. Flaw locations and orientations must be selected from the choices shown in paragraph (b)(2)(xv)(K)(*4*) of this section, Table VIII-S7-1—Modified, with the exception that flaws in the outer eighty-five (85) percent of the weld need not be perpendicular to the weld. There may be no more than two flaws from each category, and at least one subsurface flaw must be included.

(*ii*) For length sizing, a minimum of four flaws as in § 50.55a(b)(2)(xv)(K)(*1*)(*i*) must be included in the test set. The length sizing results must be added to the results of combined Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII. The combined results must meet the acceptance standards contained in § 50.55a(b)(2)(xv)(E)(*3*).

(*iii*) For depth sizing, a minimum of four flaws as in § 50.55a(b)(2)(xv)(K)(*1*)(*i*) must be included in the test set. Their depths must be distributed over the ranges of Supplement 4, Paragraph 1.1, to Appendix VIII, for the inner 15 percent of the wall thickness and Supplement 6, Paragraph 1.1, to Appendix VIII, for the remainder of the wall thickness. The depth sizing results must be combined with the sizing results from Supplement 4 to Appendix VIII for the inner 15 percent and to Supplement 6 to Appendix VIII for the remainder of the wall thickness. The combined results must meet the depth sizing acceptance criteria contained in §§ 50.55a(b)(2)(xv)(C)(*1*), 50.55a(b)(2)(xv)(E)(*1*), and 50.55a(b)(2)(xv)(F)(*3*).

(*2*) For examination of reactor pressure vessel nozzle-to-vessel welds conducted from the inside of the vessel,

(*i*) The clad to base metal interface and the adjacent examination volume to a minimum depth of 15 percent T (measured from the clad to base metal interface) must be examined from four orthogonal directions using a procedure and personnel qualified in accordance with Supplement 4 to Appendix VIII as modified by §§ 50.55a(b)(2)(xv)(B) and 50.55a(b)(2)(xv)(C).

(*ii*) When the examination volume defined in § 50.55a(b)(2)(xv)(K)(*2*)(*i*) cannot be effectively examined in all four directions, the examination must be augmented by examination from the nozzle bore using a procedure and personnel qualified in accordance with § 50.55a(b)(2)(xv)(K)(*1*).

(*iii*) The remainder of the examination volume not covered by § 50.55a(b)(*2*)(xv)(K)(*2*)(*ii*) or a combination of § 50.55a(b)(2)(xv)(K)(*2*)(i) and § 50.55a(b)(2)(xv)(K)(*2*)(*ii*), must be examined from the nozzle bore using a procedure and personnel qualified in accordance with § 50.55a(b)(2)(xv)(K)(*1*), or from the vessel shell using a procedure and personnel qualified for single sided examination in accordance with Supplement 6 to Appendix VIII, as modified by §§ 50.55a(b)(2)(xv)(D), 50.55a(b)(2)(xv)(E), 50.55a(b)(2)(xv)(F), and 50.55a(b)(2)(xv)(G).

(*3*) For examination of reactor pressure vessel nozzle-to-shell welds conducted from the outside of the vessel,

(*i*) The clad to base metal interface and the adjacent metal to a depth of 15 percent T, (measured from the clad to base metal interface) must be examined from one radial and two opposing circumferential directions using a procedure and personnel qualified in accordance with Supplement 4 to Appendix VIII, as modified by §§ 50.55a(b)(2)(xv)(B) and 50.55a(b)(2)(xv)(C), for examinations performed in the radial direction, and Supplement 5 to Appendix VIII, as modified by § 50.55a(b)(2)(xv)(J), for examinations performed in the circumferential direction.

(*ii*) The examination volume not addressed by § 50.55a(b)(2)(xv)(K)(*3*)(*i*) must be examined in a minimum of one radial direction using a procedure and personnel qualified for single sided examination in accordance with Supplement 6 to Appendix VIII, as modified by §§ 50.55a(b)(2)(xv)(D), 50.55a(b)(2)(xv)(E), 50.55a(b)(2)(xv)(F), and 50.55a(b)(2)(xv)(G).

(*4*) Table VIII-S7-1, ''Flaw Locations and Orientations,'' Supplement 7 to Appendix VIII, is modified as follows:

TABLE VIII-S7-1—MODIFIED

Flaw Locations and Orientations		
	Parallel to weld	Perpendicular to weld
Inner 15 percent	X	X
OD Surface	X	
Subsurface	X	

(L) As a modification to the requirements of Supplement 8, Subparagraph 1.1(c), to Appendix VIII, notches may be located within one diameter of each end of the bolt or stud.

(M) When implementing Supplement 12 to Appendix VIII, only the provisions related to the coordinated implementation of Supplement 3 to Supplement 2 performance demonstrations are to be applied.

(xvi) *Appendix VIII single side ferritic vessel and piping and stainless steel piping examination*.

(A) Examinations performed from one side of a ferritic vessel weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single side examinations. To demonstrate equivalency to two sided examinations, the demonstration must be performed to the requirements of Appendix VIII as modified by this paragraph and §§ 50.55a(b)(2)(xv) (B) through (G), on specimens containing flaws with non-optimum sound energy reflecting characteristics or flaws similar to those in the vessel being examined.

(B) Examinations performed from one side of a ferritic or stainless steel pipe weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single side examinations. To demonstrate equivalency to two sided examinations, the demonstration must be performed to the requirements of Appendix VIII as modified by this paragraph and § 50.55a(b)(2)(xv)(A).

(xvii) *Reconciliation of Quality Requirements*. When purchasing replacement items, in addition to the reconciliation provisions of IWA–4200, 1995 Addenda through 1998 Edition, the replacement items must be purchased, to the extent necessary, in accordance with the licensee's quality assurance program description required by 10 CFR 50.34(b)(6)(ii).

(xviii) *Certification of NDE personnel*. (A) Level I and II nondestructive examination personnel shall be recertified on a 3-year interval in lieu of the 5-year interval specified in the 1997 Addenda and 1998 Edition of IWA–2314, and IWA–2314(a) and IWA–2314(b) of the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(B) Paragraph IWA–2316 of the 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, may only be used to qualify personnel that observe for leakage during system leakage and hydrostatic tests conducted in accordance with IWA–5211(a) and (b), 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(C) When qualifying visual examination personnel for VT–3 visual examinations under paragraph IWA–2317 of the 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, the proficiency of the training must be demonstrated by administering an initial qualification examination and administering subsequent examinations on a 3-year interval.

(xix) *Substitution of alternative methods*. The provisions for the substitution of alternative examination methods, a combination of methods, or newly developed techniques in the 1997 Addenda of IWA–2240 must be applied. The provisions in IWA–2240, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, are not approved for use. The provisions in IWA–4520(c), 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, allowing the substitution of alternative examination methods, a combination of methods, or newly developed techniques for the methods specified in the Construction Code are not approved for use.

(xx) *System leakage tests*. When performing system leakage tests in accordance IWA–5213(a), 1997 through 2002 Addenda, a 10-minute hold time after attaining test pressure is required for Class 2 and Class 3 components that are not in use during normal operating conditions, and no hold time is required for the remaining Class 2 and Class 3 components provided that the system has been in operation for at least 4 hours for insulated components or 10 minutes for uninsulated components.

(xxi) *Table IWB–2500–1 examination requirements*. (A) The provisions of Table IWB–2500–1, Examination Category B-D, Full Penetration Welded Nozzles in Vessels, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B) in the 1998 Edition must be applied when using the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section. A visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria in Table IWB–3512–1, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, may be performed in place of an ultrasonic examination.

(B) The provisions of Table IWB–2500–1, Examination Category B-G-2, Item B7.80, that are in the 1995 Edition are applicable only to reused bolting when using the 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(xxii) *Surface Examination*. The use of the provision in IWA–2220, ''Surface Examination,'' of Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, that allow use of an ultrasonic examination method is prohibited.

(xxiii) *Evaluation of Thermally Cut Surfaces*. The use of the provisions for eliminating mechanical processing of thermally cut surfaces in IWA–4461.4.2 of Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section are prohibited.

(xxiv) *Incorporation of the Performance Demonstration Initiative and Addition of Ultrasonic Examination Criteria*. The use of Appendix VIII and the supplements to Appendix VIII and Article I–3000 of Section XI of the ASME BPV Code, 2002 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, is prohibited.

(xxv) *Mitigation of Defects by Modification*. The use of the provisions in IWA–4340, ''Mitigation of Defects by Modification,'' Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section are prohibited.

(xxvi) *Pressure Testing Class 1, 2, and 3 Mechanical Joints*. The repair and replacement activity provisions in IWA–4540(c) of the 1998 Edition of Section XI for pressure testing Class 1, 2, and 3 mechanical joints must be applied when using the 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(xxvii) *Removal of Insulation*. When performing visual examinations in accordance with IWA–5242 of Section XI, 2003 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of the section, insulation must be removed from 17–4 PH or 410 stainless steel studs or bolts aged at a temperature below 1100 °F or having a Rockwell Method C hardness value above 30, and from A–286 stainless steel studs or bolts preloaded to 100,000 pounds per square inch or higher.

(3) As used in this section, references to the OM Code refer to the ASME *Code for Operation and Maintenance of Nuclear Power Plants*, and include the 1995 Edition through the 2003 Addenda subject to the following limitations and modifications:

(i) *Quality Assurance*. When applying editions and addenda of the OM Code, the requirements of NQA–1, ''Quality Assurance Requirements for Nuclear Facilities,'' 1979 Addenda, are acceptable as permitted by ISTA 1.4 of the 1995 Edition through 1997 Addenda or ISTA–1500 of the 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, provided the licensee uses its 10 CFR Part 50, Appendix B, quality assurance program in conjunction with the OM Code requirements. Commitments contained in the licensee's quality assurance program description that are more stringent than those contained in NQA–1 govern OM Code activities. If NQA–1 and the OM Code do not address the commitments contained in the licensee's Appendix B quality assurance program description, the commitments must be applied to OM Code activities.

(ii) *Motor-Operated Valve testing.* Licensees shall comply with the provisions for testing motor-operated valves in OM Code ISTC 4.2, 1995 Edition with the 1996 and 1997 Addenda, or ISTC–3500, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, and shall establish a program to ensure that motor-operated valves continue to be capable of performing their design basis safety functions.

(iii) [Reserved]

(iv) *Appendix II.* Licensees applying Appendix II, ''Check Valve Condition Monitoring Program,'' of the OM Code, 1995 Edition with the 1996 and 1997 Addenda, shall satisfy the requirements of (b)(3)(iv)(A), (b)(3)(iv)(B), and (b)(3)(iv)(C) of this section. Licensees applying Appendix II, 1998 Edition through the 2002 Addenda, shall satisfy the requirements of (b)(3)(iv)(A), (b)(3)(iv)(B), and (b)(3)(iv)(D) of this section.

(A) Valve opening and closing functions must be demonstrated when flow testing or examination methods (nonintrusive, or disassembly and inspection) are used;

(B) The initial interval for tests and associated examinations may not exceed two fuel cycles or 3 years, whichever is longer; any extension of this interval may not exceed one fuel cycle per extension with the maximum interval not to exceed 10 years; trending and evaluation of existing data must be used to reduce or extend the time interval between tests.

(C) If the Appendix II condition monitoring program is discontinued, then the requirements of ISTC 4.5.1 through 4.5.4 must be implemented.

(D) The provisions of ISTC–3510, ISTC–3520, and ISTC–3540 in addition to ISTC–5221 must be implemented if the Appendix II condition monitoring program is discontinued.

(v) *Subsection ISTD.* Article IWF–5000, ''Inservice Inspection Requirements for Snubbers,'' of the ASME BPV Code, Section XI, provides inservice inspection requirements for examinations and tests of snubbers at nuclear power plants. Licensees may use Subsection ISTD, ''Inservice Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants,'' ASME OM Code, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, in place of the requirements for snubbers in Section XI, IWF–5200(a) and (b) and IWF–5300(a) and (b), by making appropriate changes to their technical specifications or licensee-controlled documents. Preservice and inservice examinations must be performed using the VT–3 visual examination method described in IWA–2213.

(vi) *Exercise interval for manual valves.* Manual valves must be exercised on a 2-year interval rather that the 5-year interval specified in paragraph ISTC–3540 of the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, provided that adverse conditions do not require more frequent testing.

(4) *Design, Fabrication, and Materials Code Cases.* Licensees may apply the ASME Boiler and Pressure Vessel Code cases listed in NRC Regulatory Guide 1.84, Revision 33, without prior NRC approval subject to the following:

(i) When an applicant or licensee initially applies a listed Code case, the applicant or licensee shall apply the most recent version of that Code case incorporated by reference in this paragraph.

(ii) If an applicant or licensee has previously applied a Code case and a later version of the Code case is incorporated by reference in this paragraph, the applicant or licensee may continue to apply the previous version of the Code case as authorized, or may apply the later version of the Code case, including any NRC-specified conditions placed on its use, until it updates its Code of Record for the component being constructed.

(iii) Application of an annulled Code case is prohibited unless an applicant or licensee applied the listed Code case prior to it being listed as annulled in Regulatory Guide 1.84. If an applicant or licensee has applied a listed Code case that is later listed as annulled in Regulatory Guide 1.84, the applicant or licensee may continue to apply the Code case until it updates its Code of Record for the component being constructed.

(5) *Inservice Inspection Code Cases.* Licensees may apply the ASME Boiler and Pressure Vessel Code cases listed in Regulatory Guide 1.147 through Revision 14, without prior NRC approval subject to the following:

(i) When a licensee initially applies a listed Code case, the licensee shall apply the most recent version of that Code case incorporated by reference in this paragraph.

(ii) If a licensee has previously applied a Code case and a later version of the Code case is incorporated by reference in this paragraph, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code case as authorized or may apply the later version of the Code case, including any NRC-specified conditions placed on its use.

(iii) Application of an annulled Code case is prohibited unless a licensee previously applied the listed Code case prior to it being listed as annulled in Regulatory Guide 1.147. Any Code case listed as annulled in any Revision of Regulatory Guide 1.147 which a licensee has applied prior to it being listed as annulled, may continue to be applied by that licensee to the end of the 120-month interval in which the Code case was implemented.

(6) *Operation and Maintenance of Nuclear Power Plants Code Cases.* Licensees may apply the ASME Operation and Maintenance Nuclear Power Plants Code cases listed in Regulatory Guide 1.192 without prior NRC approval subject to the following:

(i) When a licensee initially applies a listed Code case, the licensee shall apply the most recent version of that Code case incorporated by reference in this paragraph.

(ii) If a licensee has previously applied a Code case and a later version of the Code case is incorporated by reference in this paragraph, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code case as authorized or may apply the later version of the Code case, including any NRC-specified conditions placed on its use.

(iii) Application of an annulled Code case is prohibited unless a licensee previously applied the listed Code case prior to it being listed as annulled in Regulatory Guide 1.192. If a licensee has applied a listed Code case that is later listed as annulled in Regulatory Guide 1.192, the licensee may continue to apply the Code case to the end of the current 120-month interval.

(c) *Reactor coolant pressure boundary.* (1) Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III[4,5] of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.

(2) Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary as defined in § 50.2 need not meet the requirements of paragraph (c)(1) of this section, *Provided:*

(i) In the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system; or

(ii) The component is or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

(3) The Code edition, addenda, and optional ASME Code cases to be applied to components of the reactor

See footnotes at end of section.

coolant pressure boundary must be determined by the provisions of paragraph NCA–1140, Subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but—

(i) the edition and addenda applied to a component must be those which are incorporated by reference in paragraph (b)(1) of this section,

(ii) the ASME Code provisions applied to the pressure vessel may be dated no earlier than the Summer 1972 Addenda of the 1971 edition,

(iii) the ASME Code provisions applied to piping, pumps, and valves may be dated no earlier than the Winter 1972 Addenda of the 1971 edition, and

(iv) The optional Code cases applied to a component must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (b) of this section.

(4) For a nuclear power plant whose construction permit was issued prior to May 14, 1984 the applicable Code Edition and Addenda for a component of the reactor coolant pressure boundary continue to be that Code Edition and Addenda that were required by Commission regulations for such component at the time of issuance of the construction permit.

(d) *Quality Group B components.* (1) For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group B[9] must meet the requirements for Class 2 Components in Section III of the ASME Boiler and Pressure Vessel Code.

(2) The Code edition, addenda, and optional ASME Code cases to be applied to the systems and components identified in paragraph (d)(1) of this section must be determined by the rules of paragraph NCA–1140, Subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but—

(i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section,

(ii) the ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and

(iii) The optional Code cases must be those listed in the NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (b) of this section.

(e) *Quality Group C components.* (1) For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group C[9] must meet the requirements for Class 3 components in Section III of the ASME Boiler and Pressure Vessel Code.

(2) The Code edition, addenda, and optional ASME Code cases to be applied to the systems and components identified in paragraph (e)(1) of this section must be determined by the rules of paragraph NCA–1140, subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but—

(i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section,

(ii) the ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and

(iii) The optional Code cases must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (b) of this section.

(f) *Inservice testing requirements.* Requirements for inservice inspection of Class 1, Class 2, Class 3, Class MC, and Class CC components (including their supports) are located in § 50.55a(g).

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, pumps and valves must meet the test requirements of paragraphs (f)(4) and (f)(5) of this section to the extent practical. Pumps and valves which are part of the reactor coolant pressure boundary must meet the requirements applicable to components which are classified as ASME Code Class 1. Other pumps and valves that perform a function to shut down the reactor or maintain the reactor in a safe shutdown condition, mitigate the consequences of an accident, or provide overpressure protection for safety-related systems (in meeting the requirements of the 1986 Edition, or later, of the Boiler and Pressure Vessel or OM Code) must meet the test requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, pumps and valves which are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice tests for operational readiness set forth in editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, or 1.192 that are incorporated by reference in paragraph (b) of this section) in effect six months before the date of issuance of the construction permit. The pumps and valves may meet the inservice test requirements set forth in subsequent editions of this Code and addenda which are incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, or 1.192 that are incorporated by reference in paragraph (b) of this section), subject to the applicable limitations and modifications listed therein.

(3) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after July 1, 1974:

(i)–(ii) [Reserved]

(iii)(A) Pumps and valves, in facilities whose construction permit was issued before November 22, 1999, which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in the editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, or 1.192 that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(B) Pumps and valves, in facilities whose construction permit is issued on or after November 22, 1999, which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code (or the optional ASME Code cases listed in NRC Regulatory Guide 1.192 that is incorporated by reference in paragraph (b) of this section) referenced in paragraph (b)(3) of this section at the time the construction permit is issued.

(iv)(A) Pumps and valves, in facilities whose construction permit was issued before November 22, 1999, which are classified as ASME Code Class 2 and Class 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in the editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(v) All pumps and valves may meet the test requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section.

(4) Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves which are classified as ASME Code Class 1, Class 2 and Class 3 must meet the inservice test requirements, except design and access provisions, set forth in the ASME OM Code and addenda that become effective subsequent to editions and addenda specified in paragraphs (f)(2) and (f)(3) of this section and that are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components.

(i) Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months before the date of issuance of the operating license (or the optional ASME Code cases listed in NRC Regulatory Guide 1.192 that is incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during successive 120-month intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120-month interval (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, or 1.192 that are incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section.

(iii) [Reserved]

(iv) Inservice tests of pumps and valves may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section, and subject to Commission approval. Portions of editions or addenda may be used provided that all related requirements of the respective editions or addenda are met.

(5)(i) The inservice test program for a boiling or pressurized water-cooled nuclear power facility must be revised by the licensee, as necessary, to meet the requirements of paragraph (f)(4) of this section.

(ii) If a revised inservice test program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specifications to conform the technical specification to the revised program. The licensee shall submit this application, as specified in § 50.4, at least 6 months before the start of the period during which the provisions become applicable, as determined by paragraph (f)(4) of this section.

(iii) If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in § 50.4, information to support the determination.

(iv) Where a pump or valve test requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice test program as permitted by paragraph (f)(4) of this section, the basis for this determination must be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the test is determined to be impractical.

(6)(i) The Commission will evaluate determinations under paragraph (f)(5) of this section that code requirements are impractical. The Commission may grant relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

(ii) The Commission may require the licensee to follow an augmented inservice test program for pumps and valves for which the Commission deems that added assurance of operational readiness is necessary.

(g) *Inservice inspection requirements.* Requirements for inservice testing of Class 1, Class 2, and Class 3 pumps and valves are located in § 50.55a(f).

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued before January 1, 1971, components (including supports) must meet the requirements of paragraphs (g)(4) and (g)(5) of this section to the extent practical. Components which are part of the reactor

coolant pressure boundary and their supports must meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves, and their supports must meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components (including supports) which are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice examination of such components (including supports) and must meet the preservice examination requirements set forth in editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, that are incorporated by reference in paragraph (b) of this section) in effect six months before the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of this Code which are incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, that are incorporated by reference in paragraph (b) of this section), subject to the applicable limitations and modifications.

(3) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after July 1, 1974:

(i) Components (including supports) which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice examination of these components and must meet the preservice examination requirements set forth in the editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular component.

(ii) Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 must be designed and be provided with access to enable the performance of inservice examination of these components and must meet the preservice examination requirements set forth in the editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular component.

(iii)–(iv) [Reserved]

(v) All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.

(4) Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2 and Class 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this section and that are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components. Components which are classified as Class MC pressure retaining components and their integral attachments, and components which are classified as Class CC pressure retaining components and their integral attachments must meet the requirements, except design and access provisions and preservice examination requirements,

set forth in Section XI of the ASME Boiler and Pressure Vessel Code and Addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitation listed in paragraph (b)(2)(vi) of this section and the modifications listed in paragraphs (b)(2)(viii) and (b)(2)(ix) of this section, to the extent practical within the limitation of design, geometry and materials of construction of the components.

(i) Inservice examinations of components and system pressure tests conducted during the initial 120-month inspection interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months before the date of issuance of the operating license (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, that are incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120-month inspection interval (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, that are incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section.

(iii) Licensees may, but are not required to, perform the surface examinations of High Pressure Safety Injection Systems specified in Table IWB–2500–1, Examination Category B-J, Item Numbers B9.20, B9.21, and B9.22.

(iv) Inservice examination of components and system pressure tests may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section, and subject to Commission approval. Portions of editions or addenda may be used provided that all related requirements of the respective editions or addenda are met.

(v) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued after January 1, 1956:

(A) Metal containment pressure retaining components and their integral attachments must meet the inservice inspection, repair, and replacement requirements applicable to components which are classified as ASME Code Class MC;

(B) Metallic shell and penetration liners which are pressure retaining components and their integral attachments in concrete containments must meet the inservice inspection, repair, and replacement requirements applicable to components which are classified as ASME Code Class MC; and

(C) Concrete containment pressure retaining components and their integral attachments, and the post-tensioning systems of concrete containments must meet the inservice inspection, repair, and replacement requirements applicable to components which are classified as ASME Code Class CC.

(5)(i) The inservice inspection program for a boiling or pressurized water-cooled nuclear power facility must be revised by the licensee, as necessary, to meet the requirements of paragraph (g)(4) of this section.

(ii) If a revised inservice inspection program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specifications to conform the technical specification to the revised program. The licensee shall submit this application, as specified in § 50.4, at least six months before the start of the period during which the provisions become applicable, as determined by paragraph (g)(4) of this section.

(iii) If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in

See footnotes at end of section.

§ 50.4, information to support the determinations.

(iv) Where an examination requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice inspection program as permitted by paragraph (g)(4) of this section, the basis for this determination must be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the examination is determined to be impractical.

(6)(i) The Commission will evaluate determinations under paragraph (g)(5) of this section that code requirements are impractical. The Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

(ii) The Commission may require the licensee to follow an augmented inservice inspection program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.

(A) Augmented examination of reactor vessel.

(*1*) All previously granted reliefs under § 50.55a to licensees for the extent of volumetric examination of reactor vessel shell welds specified in Item B1.10 of Examination Category B-A, ''Pressure Retaining Welds in Reactor Vessel,'' in Table IWB–2500–1 of subsection IWB in applicable edition and addenda of section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, during the inservice inspection interval in effect on September 8, 1992 are hereby revoked, subject to the specific modification in § 50.55a(g)(6)(ii)(A)(*3*)(*iv*) for licensees that defer the augmented examination in accordance with § 50.55a(g)(6)(ii)(A)(*3*).

(*2*) All licensees shall augment their reactor vessel examination by implementing once, as part of the inservice inspection interval in effect on September 8, 1992, the examination requirements for reactor vessel shell welds specified in Item B1.10 of Examination Category B-A, ''Pressure Retaining Welds in Reactor Vessel,'' in Table IWB–2500–1 of subsection IWB of the 1989 Edition of section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, subject to the conditions specified in § 50.55a(g)(6)(ii)(A) (*3*) and (*4*). The augmented examination, when not deferred in accordance with the provisions of § 50.55a(g)(6)(ii)(A)(*3*), shall be performed in accordance with the related procedures specified in the section XI edition and addenda applicable to the inservice inspection interval in effect on September 8, 1992, and may be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect on September 8, 1992. For the purpose of this augmented examination, ''essentially 100% as used in Table IWB–2500–1 means more than 90 percent of the examination volume of each weld, where the reduction in coverage is due to interference by another component, or part geometry.

(*3*) Licensees with fewer than 40 months remaining in the inservice inspection interval in effect on September 8, 1992 may defer the augmented reactor vessel examination specified in § 50.55a(g)(6)(ii)(A)(*2*) to the first period of the next inspection interval under the following conditions:

(*i*) The deferred augmented examination may not be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect on September 8, 1992.

(*ii*) The deferred augmented examination may be used as a substitute for the reactor vessel shell weld examination normally scheduled for the inspection interval in which the deferred examination is performed.

(*iii*) If the deferred augmented examination is used as a substitute for the normally scheduled reactor vessel shell weld examination, subsequent reactor vessel shell weld examinations must be performed during the first period of successive inspection intervals.

(*iv*) Licensees that defer the augmented examination, as permitted herein, may retain all previously granted reliefs that otherwise would be revoked by § 50.55a(g)(6)(ii)(A)(*1*) for the inservice inspection interval in effect on September 8, 1992.

(*v*) Licensees with fewer than 40 months remaining in the inservice inspection interval in effect on September 8, 1992 may extend that interval in accordance with the provisions of section XI (1989 Edition) IWA–2430(d) for the purpose of implementing the augmented examination during that interval.

(*vi*) The deferred augmented examination shall be performed in accordance with the related procedures specified in the section XI edition and addenda applicable to the inspection interval in which the augmented examination is performed.

(*4*) The requirement for augmented examination of the reactor vessel may be satisfied by an examination of essentially 100 percent of the reactor vessel shell welds specified in § 50.55a(g)(6)(ii)(A)(*2*) that has been completed, or is scheduled for implementation with a written commitment, or is required by § 50.55a(g)(4)(i), during the inservice inspection interval in effect on September 8, 1992.

(*5*) Licensees that make a determination that they are unable to completely satisfy the requirements for the augmented reactor vessel shell weld examination specified in § 50.55a(g)(6)(ii)(A) shall submit information to the Commission to support the determination and shall propose an alternative to the examination requirements that would provide an acceptable level of quality and safety. The licensee may use the proposed alternative when authorized by the Director of the Office of Nuclear Reactor Regulation.

(B) Licensees do not have to submit to the NRC staff for approval of their containment inservice inspection programs which were developed to satisfy the requirements of Subsection IWE and Subsection IWL with specified modifications and limitations. The program elements and the required documentation must be maintained on site for audit.

(C) *Implementation of Appendix VIII to Section XI.* (*1*) Appendix VIII and the supplements to Appendix VIII to Section XI, Division 1, 1995 Edition with the 1996 Addenda of the ASME Boiler and Pressure Vessel Code must be implemented in accordance with the following schedule: Appendix VIII and Supplements 1, 2, 3, and 8—May 22, 2000; Supplements 4 and 6—November 22, 2000; Supplement 11—November 22, 2001; and Supplements 5, 7, and 10—November 22, 2002.

(*2*) Licensees implementing the 1989 Edition and earlier editions and addenda of IWA–2232 of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code must implement the 1995 Edition with the 1996 Addenda of Appendix VIII and the supplements to Appendix VIII of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code.

(h) Protection and safety systems. (1) IEEE Std. 603–1991, including the correction sheet dated January 30, 1995, which is referenced in paragraphs (h)(2) and (h)(3) of this section, is approved for incorporation by reference by the Director of the Office of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR Part 51. Copies of IEEE Std. 603–1991 may be purchased from the Institute of Electrical and Electronics Engineers Service Center, 445 Hoes Lane, Piscataway, NJ 08855. The standard is also available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Md; or at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202–741–6030, or go to: *http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html* IEEE Std. 279, which is referenced in paragraph (h)(2) of this section, was approved for incorporation by reference by the Director of the Office of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR Part 51. Copies of IEEE Std. 279 are also available as indicated for IEEE Std. 603–1991.

(2) Protection systems. For nuclear power plants with construction permits issued after January 1, 1971, but before

May 13, 1999, protection systems must meet the requirements stated in either IEEE Std. 279, ''Criteria for Protection Systems for Nuclear Power Generating Stations,'' or in IEEE Std. 603–1991, ''Criteria for Safety Systems for Nuclear Power Generating Stations,'' and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.

(3) Safety systems. Applications filed on or after May 13, 1999 for preliminary and final design approvals (10 CFR Part 52, Appendix O), design certifications, and construction permits, operating licenses and combined licenses that do not reference a final design approval or design certification, must meet the requirements for safety systems in IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.

Footnotes to § 50.55a:

[1–3] [Reserved]

[4] USAS and ASME Code addenda issued prior to the Winter 1977 Addenda are considered to be ''in effect'' or ''effective'' 6 months after their date of issuance *and* after they are incorporated by reference in paragraph (b) of this section. Addenda to the ASME Code issued after the Summer 1977 Addenda are considered to be ''in effect'' or ''effective'' after the date of publication of the addenda *and* after they are incorporated by reference in paragraph (b) of this section.

[5] For ASME Code Editions and Addenda issued prior to the Winter 1977 Addenda, the Code Edition and Addenda applicable to the component is governed by the order or contract date for the component, not the contract date for the nuclear energy system. For the Winter 1977 Addenda and subsequent editions and addenda the method for determining the applicable Code editions and addenda is contained in Paragraph NCA 1140 of Section III of the ASME Code.

[6–8] [Reserved]

[9] Guidance for quality group classifications of components which are to be included in the safety analysis reports pursuant to § 50.34(a) and § 50.34(b) may be found in Regulatory Guide 1.26, ''Quality Group Classifications and Standards for Water-, Steam-, and Radiological-Waste-Containing Components of Nuclear Power Plants,'' and in Section 3.2.2 of NUREG–0800, ''Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants.''

[10] Supplemental inservice inspection requirements for reactor vessel pressure heads have been imposed by Order EA–03–09 issued to licensees of pressurized water reactors. The NRC expects to develop revised supplemental inspection requirements, based in part upon a review of the initial implementation of the order, and will determine the need for incorporating the revised inspection requirements into 10 CFR 50.55a by rulemaking.

[36 FR 11424, June 12, 1971]

EDITORIAL NOTE: For FEDERAL REGISTER citations affecting § 50.55a, see the List of CFR Sections Affected, which appears in the Finding Aids section of the printed volume and on GPO Access.

§ 50.56 Conversion of construction permit to license; or amendment of license.

Upon completion of the construction or alteration of a facility, in compliance with the terms and conditions of the construction permit and subject to any necessary testing of the facility for health or safety purposes, the Commission will, in the absence of good cause shown to the contrary issue a license of the class for which the construction permit was issued or an appropriate amendment of the license, as the case may be.

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 11461, July 17, 1970]

§ 50.57 Issuance of operating license. [1]

(a)Pursuant to § 50.56, an operating license may be issued by the Commission, up to the full term authorized by § 50.51, upon finding that:

(1) Construction of the facility has been substantially completed, in conformity with the construction permit and the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(2) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

[1] The Commission may issue a provisional operating license pursuant to the regulations in this part in effect on March 30, 1970, for any facility for which a notice of hearing on an application for a provisional operating license or a notice of proposed issuance of a provisional operating license has been published on or before that date.